Development of Ignalina NPP RBMK-1500 reactor RELAP5-3D model

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1. Introduction

In 1999 a contract was signed between the Pacific Northwest National Laboratory (PNNL), which is representing US DOE in the frame of the US International nuclear safety program (INSP) and, as a part of it, the Soviet-designed reactor safety program, and the Lithuanian Energy Institute (LEI). The overall objective of this contract is to develop models for RBMK-1500 reactors for use in integrated thermal-hydraulics-neutronics calculations for the analysis of specific transients in which the neutronic response of the core is important. The specific technical goals of the contract are: 1) development of an RBMK-1500 whole-core neutron kinetics and thermal-hydraulics model for analysis of Ignalina NPP transients; 2) verification of the coupled code model using data from steady state or operational transients from Ignalina NPP; 3) analysis of benchmark problems defined for the purpose of assessing coupled neutron kinetics-thermal-hydraulics code RELAP5-3D for use in RBMK transient analysis. This paper deals with two first goals of the project.

2. Ignalina NPP RELAP5-3D model

Following the overall objective of the project Ignalina NPP RBMK-1500 reactor RELAP5-3D model was developed. Key features of the Ignalina NPP RELAP5-3D model are the following:

- Both Main Circulation Circuit (MCC) loops are modeled;
- The heat exchange between technological channels is modeled using heat exchange through the gap in graphite moderator in a gas circuit of the reactor. The transfer of heat from technological channels to Control and Protection System (CPS) channels and reflector cooling circuit occurs through the same gas circuit;
- The paths for steam removal from the drum separator (DS) are modeled explicitly, including the steam lines, safety valves, the high pressure rings, the steam feeding to the turbines and to the local consumers;
- The systems feedwater and the emergency cooling systems are modeled explicitly;
- The control rod cooling and radial reflector cooling circuit are modeled explicitly;
- The CPS logic, automatic pressure regulators and the water level in DS are modeled;
- The Accident Confinement System (ACS) is not modeled. Consequently, system inputs for the control system based on the ACS response must be user specified;
- The reactor thermal power is calculated using the nodal kinetic model of RELAP5-3D code. All structures (fuel, coolant, graphite-moderator, control rods) are included in this nodal kinetic model.

2.1. Thermal-hydraulic part of Ignalina NPP RELAP5-3D model

The nodalization scheme of the model is presented in Figure 1. The model of the MCC consists of two loops, each of which corresponds to one loop of the actual circuit. The left half in the model is simplified so that all components are incorporated among themselves. This half has one generalized Main Circulation Pump (MCP), Suction Header (SH), Pressure Header (PH), generalized Group Distribution Header (GDH), and generalized DS. In this half damage to separate elements cannot be simulated.

The right half of the MCC is modeled in finer detail, thus permitting simulation of damage to separate elements. Both loops of the circuit are connected among themselves through the steam lines. The steam lines and rings of high pressure are represented by RELAP5 pipe components "175" and "575" in the MCC model. Two drum separators of each MCC loop of MCC are combined and are described by three elements (elements " 100 "," 101 "," 102 " and " 500 "," 501 "," 502 "). Elements " 100 " and " 500 " are RELAP5 separator components that represent the DS volumes with shipped perforated plate used for steam separation. These are the volumes in which the steam separation occurs. RELAP5 branch components " 102 " and " 502 " collect the separated water, and RELAP5 branch components " 101 " and " 501 " collect the saturated steam. The downcomers are represented by RELAP5 pipe components "120" (in the left half) and "520" (in the right half). Each component simulates 24 downcomers. The suction and pressure headers are modeled by RELAP5 branch components "125", "525" and "145", "545" respectively. It is assumed in the model that three MCPs are operating in each loop of MCC. RELAP5 pump component "135" represents three, lumped MCPs of the left loop, and each MCP of the right loop is explicitly represented by RELAP5 pump components "535", "635" and "835". The bypass pipes connecting suction and pressure headers are modeled with the closed valves (valves "127" and "527").

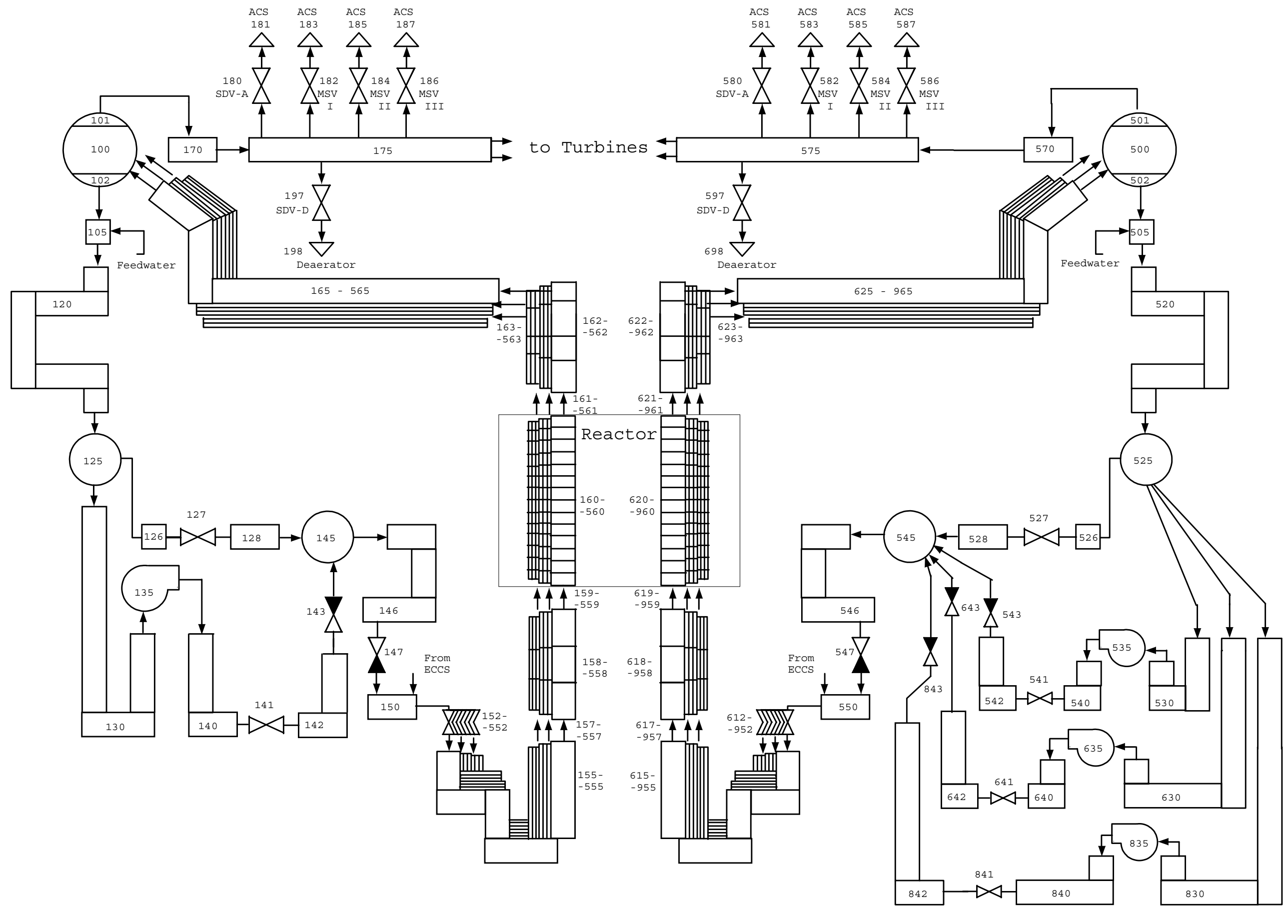

Figure 1. Ignalina NPP thermal-hydraulic model nodalization diagram

The reactor core is modeled by 14 RELAP5 pipe components, each of which represents a separate group of fuel channels (FC). Seven RELAP5 pipe components represent the 835 FC in the left loop and seven RELAP5 pipe components represent the 826 FC in the right loop. The distribution of FC in both MCC loops is shown in the Table 1.

Table 1. Modeling of equivalent fuel channels in the RBMK-1500 reactor RELAP5-3D model

Right loop of MCC		Left loop of MCC	
RELAP5-3D element	Number of FC	RELAP5-3D element	Number of FC
620	355	160	378
710	249	210	234
730	60	280	59
760	59	390	55
780	39	410	37
860	61	460	70
960	3	560	2
Total in right loop:	826	Total in left loop:	835

The group of 20 distribution headers with connecting pipelines is modeled by RELAP5 components "146 - 150" in the left MCC loop. RELAP5 components "546 - 550" model GDH with connecting pipelines in the right MCC loop of MCC. The pipelines of the water communications are connected to each GDH. In the nodalization scheme they are represented by RELAP5 components "155"-"555" and "615"-"955". Each of these components represents the quantity of pipes appropriate to the number of elements in the corresponding FC in the core. The vertical parts of the FC above the reactor core are represented by RELAP5 components "162"-"562" and "622"-"962". The pipelines of the steam-water communications connected to the vertical parts of the FC are represented by RELAP5 components "165"-"565" and "625"-"965".

Square profile 0.25 x 0.25 m graphite blocks are modeled by cylindrical elements with the equivalent cross-section area. The outer radius is R=0.141 m, which gives a cross-sectional area for the heat structure equivalent to that of the actual graphite column. The heat structure of the equivalent fuel channel simulates not only active region in the reactor core, but the top and bottom reflectors are modeled also. Each equivalent channel is modeled using 16 axial nodes of length 0.5 m each. The fuel element is modeled using eight radial nodes, five to represent the fuel pellet, one for the gap region and two for the cladding. The fuel channels and graphite columns are modeled using eight radial nodes. Two of these radial nodes are for the fuel channel wall, two for the gap and graphite rings region and four for the graphite column.

The energy that is dissipated around the MCC is evaluated by determining the energy added to the fluid in the MCPs. The use of RELAP5-3D code allows description of the heat exchange between technological channels, CPS channels and the reflector cooling circuit without using the detailed reactor gas circuit model. This heat transfer is described using the special 'Conduction Input' option. The heat transfer between 14 equivalent fuel channels, one equivalent CPS channel and one equivalent reflector cooling circuit channel are modeled. Heat exchange between graphite columns occurs along all length of equivalent channel. It is assumed that conductance of the gap between graphite columns is constant and equal 140 $W/(m^2 \cdot K)$. A view factor of the adjacent heat structure is calculated according outer surface area of graphite columns.

2.2. Nodal kinetics part of Ignalina NPP RELAP5-3D model

The RBMK-1500 reactor core has a 7.0 m fuel region and a 0.5 m reflector region above and below the fuel region. The overall height of the core region is 8.0 m. The neutronics mesh represents each rectangular graphite column as one individual stack in the radial plane. The reactor core region in the RBMK-1500 RELAP5-3D model has 32 axial nodes (0.25 m each) and 56x56 nodes (0.25 m each) in the radial plane. This mesh results in 28 axial nodes in the fuel region and 2 axial nodes in each of the top and bottom reflector region (see Figure 2). In thermal-hydraulic model of the reactor core we have 16 thermal-hydraulic meshes: 14 nodes (0.5 m each) in the fuel region and 1 node in each of the top and bottom reflector region. In this way the height of the two neutronics nodes are equal to the height of one thermal-hydraulic node.

The reactor core composition in the model is represented by 2 composition maps: one for the top and bottom reflector regions and one for the fuel region. There are 12 different compositions that are present in the reactor core layout. The summary specification of the different components of the reactor core layout are presented in Table 2. There one can find 3 types of fuel assemblies, 4 types of CPS control rods and other different instrumentation used in the reactor core.

Table 2. Summary specification of the compositions in the core loading map of the 2nd unit of Ignalina NPP (actual state of the core for 1998.11.26.)

Loading of the core	Specification of the channel in the core loading map	Number of the channels
Fuel channel with burnable absorber (2,4% enr. fuel with Erbium)	1	1170
Fuel channel (2% enr. fuel)	3	478
Fuel channel with recycled fuel (2% enr. fuel)	5	7
Radial reflector	12	436
Radial reflector cooling channel	13	156
Fission chambers and axial detectors	14	24
Experimental channel	15	1
Water column	16	5
Additional absorbers (cluster type)	18	0
Manual control rods (mod. 2091-01)	24	76
Short absorber control rods	25	40
Fast scram rods	26	24
Manual control rods (mod. 2477-01)	27	71

A model of nodal reactor kinetics is based on real state of the reactor of the 2nd Unit of Ignalina NPP, registered by ICS "TITAN" on the 26th of November 1998. Reactor core loading information was obtained from the plant as a part of the database from the main information computer system "TITAN". Besides the reactor core loading information, the database provided the following information that was used in RBMK-1500 RELAP5-3D model: insertion depth of the CPS control rods, burnup of each of the fuel assemblies, axial fuel burnup profile, coolant flowrate maps of the MCC and the CPS cooling circuit. Radial fuel assemblies burnup profile and axial relative fuel burnup profile were input into the model as user input variable.

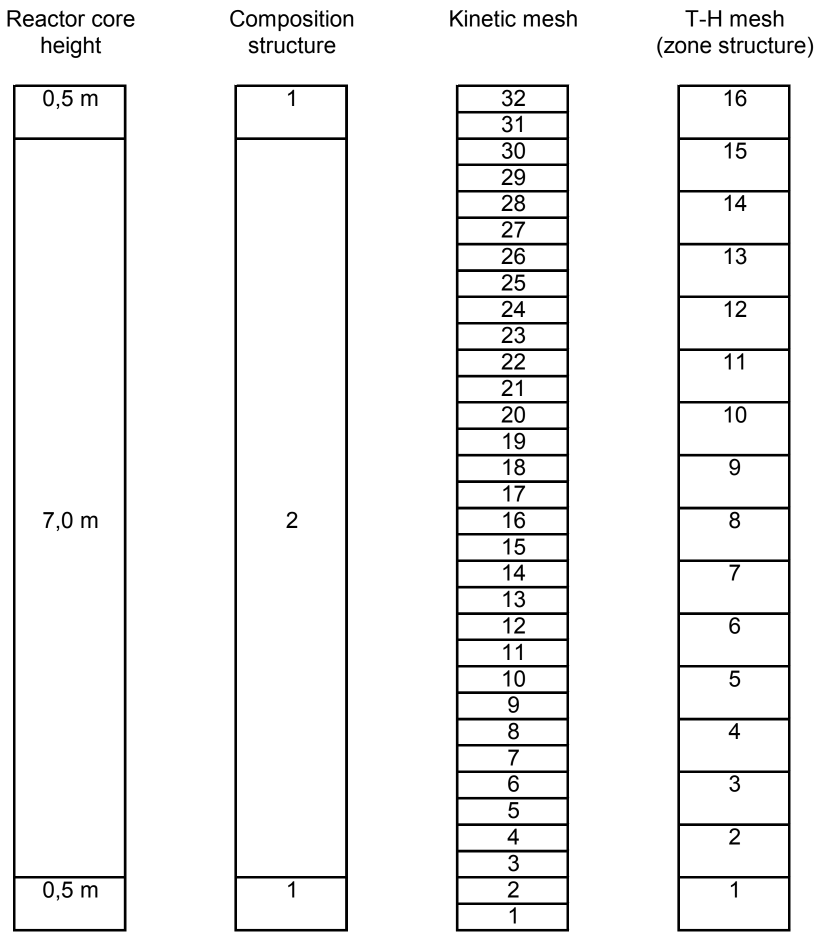

Figure 2. Kinetic mesh, thermal-hydraulic mesh and material composition structure of Ignalina NPP RBMK-1500 reactor RELAP5-3D model

Cross sections for the different compositions of the RBMK-1500 reactor core were obtained from two-group macro x-section library of the STEPAN code that was provided to us by Russian Research Center "Kurchatov Institute". X-section library includes subroutines for fuel cells, non-fuel cells and the CPS control rods. An external user subroutine interface was written that accesses the coding of the RRC "KI" x-section library subroutines at each time step of the calculation. The interface receives thermal-hydraulic and control rod position information from the RELAP5-3D code and provides input to the RRC "KI" x-section library subroutines. X-section library subroutines return the diffusion, absorption, fission and scattering x-sections for the two neutron groups. The interface then transfers the obtained x-sections to the NESTLE code kinetics solver that is part of the RELAP5-3D code.

For the fuel cells the RRC "KI" x-section library subroutines also need relative node power level to correct for the xenon radial and axial distribution in the reactor core. This relative power in the reactor core for the very first step of reactor core state calculation is set equal to 0.864 ($N_{actual}/N_{max}^{design}$, where N is reactor power) for all the kinetics nodes. For all the following steady-state calculation steps it is calculated using the equation presented below:

pow = (phi(1)*sigf1p(ixyz) + phi(2)*sigf2p(ixyz))*G*K*V*N/R,

where:

phi(i) - the neutron flux for group i in the current mesh position;
sigf1p(ixyz) - the macroscopic fission x-section for group i for the current mesh position (saved from the previous time step)
G = 200 MeV/fission;
K = 1.6021917e-13 J/MeV;
N = 1661 - number of fuel assemblies;
V = 25*25*700 = 437500 cm^3 – fuel assembly volume;
R = 4800 MW - rated power.

The equation formulated in terms of neutron group flux and fission cross-section because the flux values are available directly to the external subroutine, whereas node power is not. The STEPAN cross-section libraries use the relative power value in the calculation of the xenon relative equilibrium value. This relative xenon value distribution in the reactor core is calculated only for the steady-state case. For a transient case the relative xenon value is “frozen” and is taken as constant for each kinetics node, and is input to the kinetics in tabular form as a user variable.

Figure 3 shows the assignment of thermal-hydraulic channel groups to the radial kinetics nodes of the RBMK-1500 reactor core. As previously described, the reactor core is divided into two halves with 7 thermal-hydraulic channels per core half. There are 2 additional thermal hydraulic channels that model 1) radial reflector and radial reflector cooling channels lumped together, and 2) the CPS cooling circuit channels lumped together. Therefore, the reactor core has 14 thermal-hydraulic channels for the fuel channels and 2 thermal-hydraulic channels for the non-fuel channels. The fuel channels were divided into 7 groups according to power and coolant flowrate values. Table 3 shows the assignment of thermal-hydraulic channels to each group. The number of channels in each group varies from 2 to 378. As shown, channel groups CC11, CC12, CC21 and CC22 are located in the center of the reactor core, and the remaining groups are on the periphery. The CC18 CPS channel group is distributed evenly all through the reactor core. The CC19 channel group represent radial reflector and radial reflector cooling channels group. ‘CC’ represents the thermal-hydraulic axial mesh number. The kinetics part of the model models each fuel and non-fuel channel individually, as shown in Figure 3.

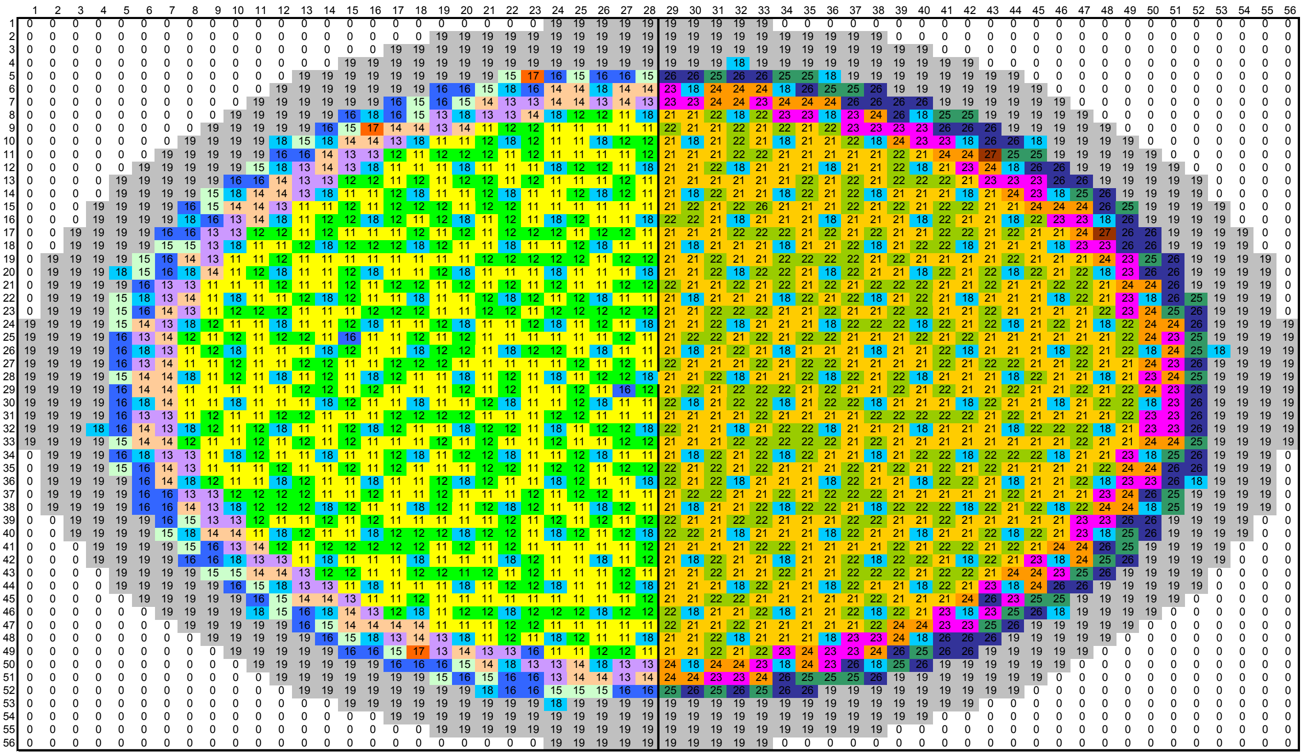

Figure 3. Nodalization scheme of the RBMK-1500 reactor core.

Another complicated part of RBMK-1500 reactor RELAP5-3D model is the CPS control rods and the CPS operation logic. All CPS 211 control rods are modeled individually, because all of them have different insertion depths into the reactor core. Four types of control rods are modeled: 2091 mod. manual control rods, 2477 mod. manual control rods, fast acting control rods and short absorber control rods. The first three types of control rods are inserted from the top of the reactor core, while the fourth type of control rods is inserted from the bottom. RELAP5-3D control variable system is used for CPS logic and CPS control rod movement modeling. Movements of the CPS control rods are controlled by the CPS logic, based on the power deviation signals coming from 127 radial detectors of the DKER-1 radial detector system. The DKER-1 detectors are modeled as having 7 sensitive elements (0.25 m each) distributed evenly over the height of the fuel region of the reactor core. Power deviation signal is based on the steady-state thermal neutron flux value in each detector location. All the detectors of the DKER-1 detector system are located in 12 local automatic control / local emergency protection (LAC/LEP) zones. In each LAC/LEP zone there is one LAC control rod and 2 LEP control rods.

Table 3. Summary specification of the thermal-hydraulic channel groups as being modeled in the RBMK-1500 reactor RELAP5-3D model

Channel group specification in the kinetics model	Channel group specification in thermal-hydraulic model	Side of the reactor	Number of channels	Average power value in a single channel, MW	Average flowrate value in a single channel, m^3/h
CC11	620	Left	355	2.95	28.2
CC21	160	Right	378	2.95	28.2
CC12	710	Left	249	2.5	26.2
CC22	210	Right	234	2.5	26.2
CC13	730	Left	60	2.4	25.1
CC23	280	Right	59	2.4	25.1
CC14	760	Left	59	1.8	21.1
CC24	390	Right	55	1.8	21.1
CC15	780	Left	39	1.6	17.5
CC25	410	Right	37	1.6	17.5
CC16	860	Left	61	1.2	15.6
CC26	460	Right	70	1.2	15.6
CC17	960	Left	3	1.8	33.5
CC27	560	Right	2	1.8	33.5
CC18	022		235		
CC19	032 & 034		592*		

* 436 channels are radial reflector channels

LAC and LEP rods move based on a certain percent deviation of the transient thermal neutron flux value from its initial value at the beginning of transient calculation. Movement of LAC rods continues until the signal that initiated their movement is no longer valid. Then the LAC rods stop moving and hold their current positions until another signal to insert or withdraw a certain control rod is received. The LAC rods move individually, depending on the power deviation signals coming from radial detectors located in their corresponding LAC zones. LEP rods can move either together with the LAC rods based on the overpower signals coming from detectors located in a certain LAC zone, or they can move separately from LAC rods based on the overpower signals coming from detectors that belong to a certain LEP zone. LEP rods move only into the core, but never out of the core. If two overpower signals are coming from DKER-1 detectors of different detector groups that belong to a single LEP zone at the same time, the AZ-6 signal in that LEP zone is initiated. The AZ-6 signal initiates the AZ-3 signal if reactor power is more that ½ of design reactor power. The AZ-3 signal causes reactor power to be reduced to ½ of design reactor power. If the AZ-6 signal is still valid when reactor power is ½ of design reactor power, reactor power is reduced further until AZ-6 signal disappears.

Modeled are one more fast controlled automatic emergency reactor power decrease mode (AZ-4), which decreases reactor power to 60% of design reactor power and which is initiated by technological parameters deviation from set-points, and two reactor emergency protection modes (FASS and AZ-1), which are triggered based on neutronic and technological parameters deviation from set-points. In FASS or AZ-1 mode all 211 control rods are inserted into the core and the reactor is shutdown in 12÷14 seconds. The difference between the two modes is, that in FASS mode fast acting scram rods are inserted in 2÷2.5 seconds into the core, while in AZ-1 mode they are inserted in 5÷7 seconds.

DKER-2 detectors are used mainly for technological purposes and are not modeled in this RBMK-1500 reactor RELAP5-3D model. DKEV detectors are used in CPS logic for emergency protection function of the CPS. There are 160 DKEV detectors that need to be modeled. But because CPS logic modeling requires a large fraction of the total available RELAP5-3D code control variable system (which has a limited number of cards), this part of CPS logic is still unavailable in RBMK-1500 reactor RELAP5-3D model. The possibility to have 160 DKEV detectors included into RELAP5-3D model will be analyzed and, if found possible (evaluating the limitation of control variable system), implemented in the nearest future.

3. Observations from recent calculations

The state-of-the-art code RELAP5-3D was originally designed for Pressurized Water Reactors. Because of the unique RBMK design, the application of this code to RBMK-1500 encountered several problems. Comparison of the calculation results with the real plant data allow to verify the suitability of the developed model for the future modeling of the processes, taking place in RBMK-1500 reactor during DBA and other different transient modes of the reactor operation.

3.1. Verification of the thermal hydraulic part of the RELAP5-3D model using actual operational events

In the process of development of the Ignalina NPP RELAP5-3D model for RBMK-1500 type reactors, comparative analyses of actual operational events are essential because this allows to establish realistic hydraulic resistances of different MCC components and realistic behavior of the controllers of the reactor systems. For this purpose the following RELAP5-3D benchmark analysis were performed:

- one and all operating MCPs trip events,
- three Main Safety Relief Valves LOCA event,
- inadvertent actuation of ECCS.

<u>Single MCP trip</u>

On May 14, 1996 one MCP at Ignalina Unit 2 was inadvertently tripped. The reactor operated at the 3400 MW thermal power prior to the event. AZ-4 signal was generated due to loss of power to the MCP. The CPS rods started to move. The turbine generator (which before the accident operated in the power control mode) switched from power control mode to drum separators pressure maintenance mode. As the flow through the pump dropped to zero (after about 5 seconds from the beginning of the accident) the check valve (downstream of this MCP) closed, preventing a reverse flow through the tripped pump. After one MCP trip, the throughput of two running pumps increased by ~1500 m^3/h. However, the total coolant flow through the affected loop decreased from 23500 m^3/h to 19000 m^3/h. Comparison between calculated flow rates obtained by RELAP5-3D model and real measured data is presented in Figure 4. The figure shows a favorable coincidence of MCP throughputs and coolant flow rate through the reactor.

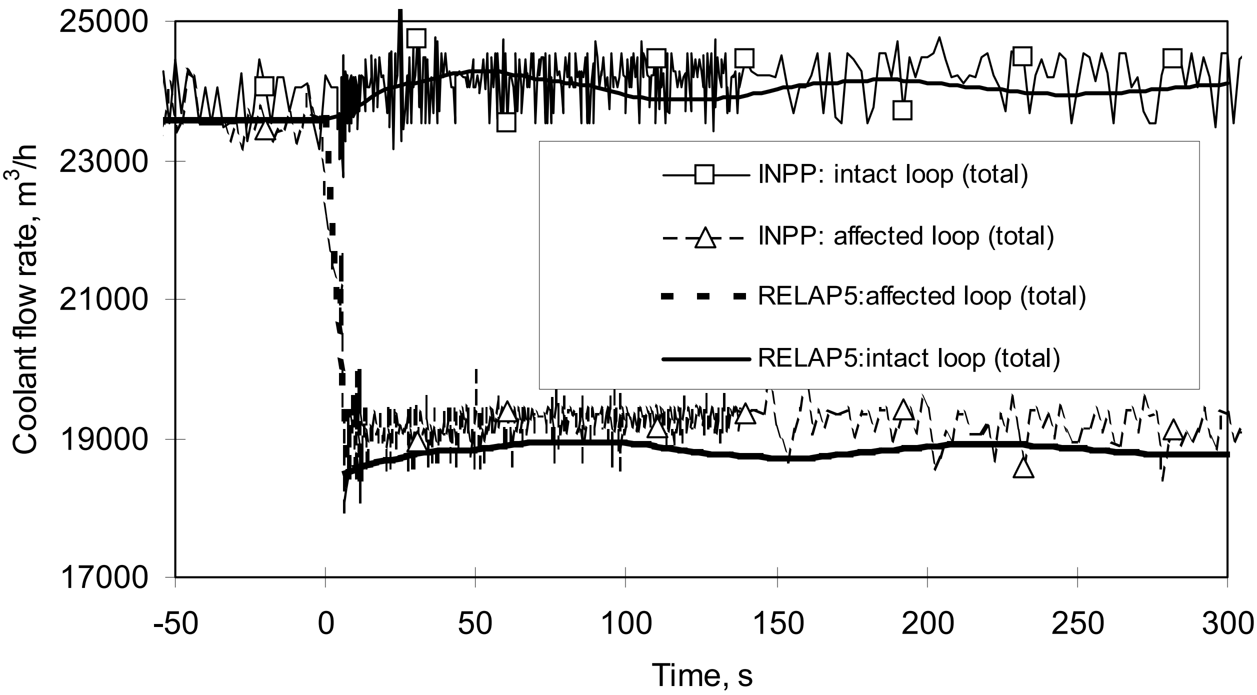

Figure 4. Single MCP trip. Coolant flow rate through reactor

<u>One MCP trip with failure of check valve</u>

The similar event took place on January 23, 1998. In this case, the one MCP trip with failure of check valve occurred. The reactor operated at the 3700 MW thermal power prior to the event. After the pump trip each of the two operating pumps (in the affected loop) increased its throughput from 7750 m^3/h to 10100 m^3/h. The increased pump throughput effectively compensated for the reversed flow through the failed valve. The net flow supplied to the affected core side decreased from 2300 m3/h to 15500 m^3/h. A comparison of measured data and calculated flows through fuel channels is shown in Figure 5.

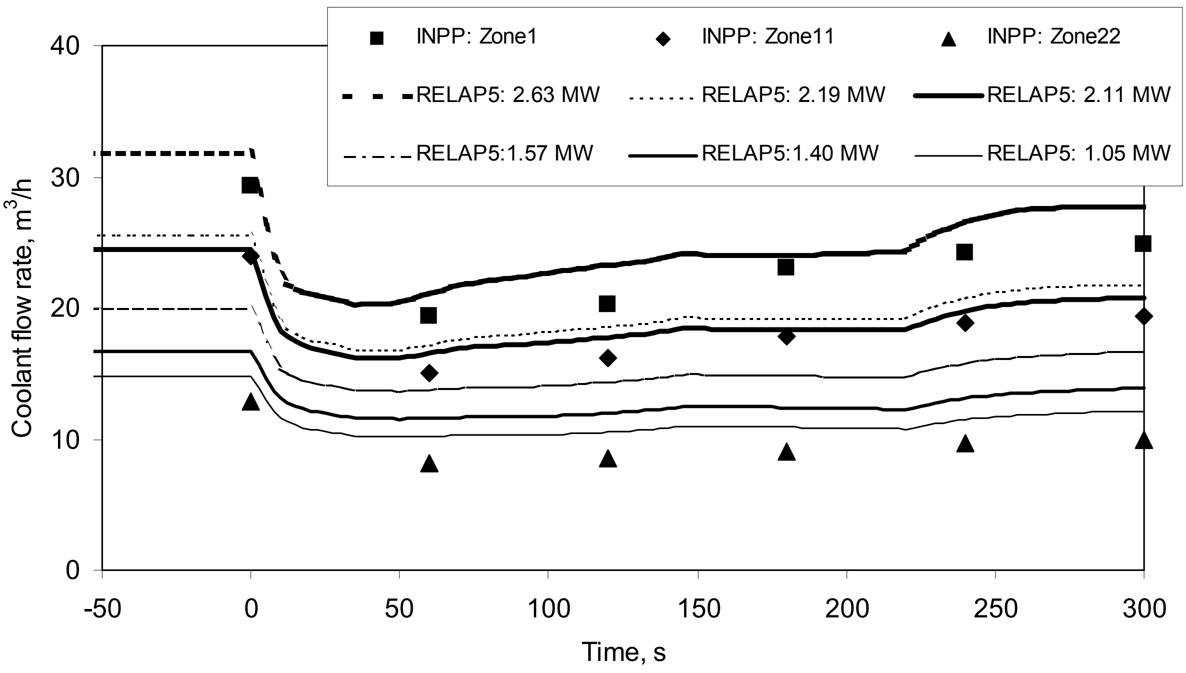

Figure 5. One MCP trip with failure of check valve. Flow rates in separate fuel channels

Loss-of-all-MCPs transient

In order to benchmark the RELAP5-3D model transient analysis calculations were performed for the loss-of-all-MCPs event. This is an actual transient that occurred at Ignalina NPP. On March 26, 1986 all six operating MCPs at Ignalina Unit 1 were tripped simultaneously. Before this event reactor operate at thermal power level of 4650 MW. In response to multiple pump trip, an emergency protection signal AZ-1 was generated and reactor was shutdown. The MCC flow decreased in response to the MCPs cost-down. Long-term flow was due to natural circulation in the MCC. Analysis results are compared against the plant data. Flow rates through individual channels are presented in Figure 6. Flow rate through the fuel channel decreases due to loss of forced circulation by the MCPs. The coast-down occurred within approximately 40 seconds from the beginning of transient. The flow coast-down to a long-term natural circulation at flow rate equal approximately 15 percent of the initial flow. The analysis results agree well with the measured flow rates.

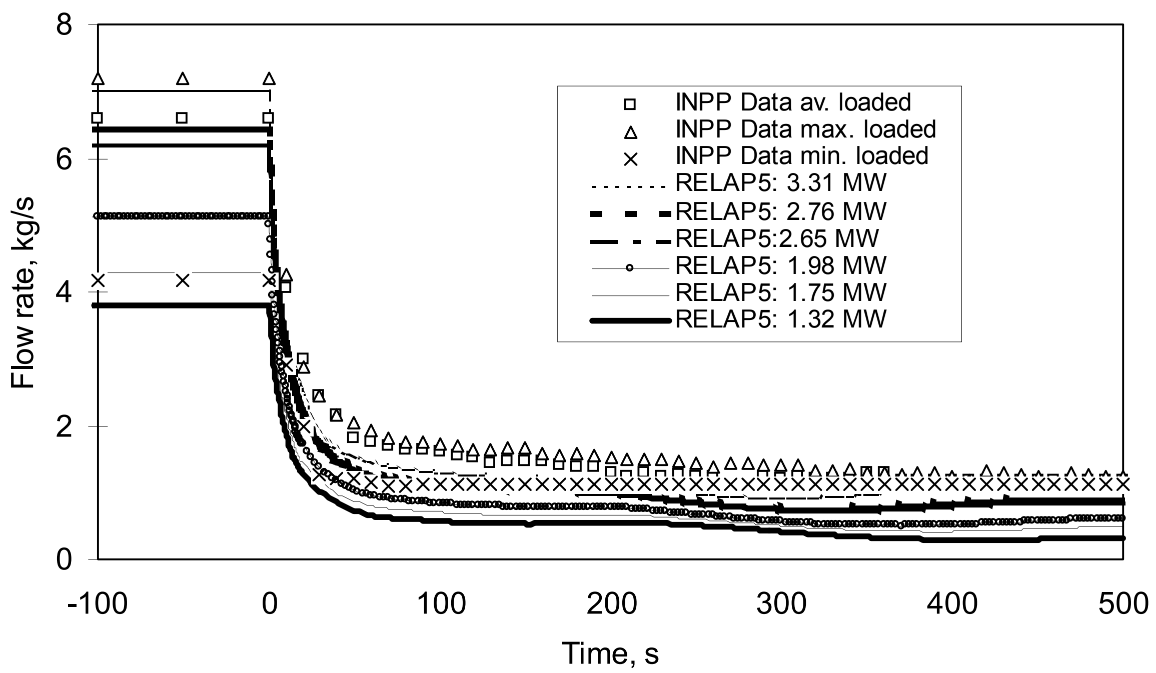

Figure 6. Loss-of-all-MCPs transient. Flow rate through individual channels

Three MSRVs LOCA Event

Three Main Safety Relief Valves (MSRVs) of third group were spuriously opened at the Ignalina NPP Unit 1 at November 27, 1986. Before the event, reactor was operated at thermal power level of 4350 MW. Three MSRVs spuriously opened at 79 s and closed at 261.5 s from the beginning of the accident. Turbine Control Valve of the turbine, which operates in the 'pressure following' logic, immediately after three MSRVs opening, decreases steam flow rate until the value, which is equal to amount of steam through three MSRVs. The total steam flow rate to turbines and for local consumers is shown in Figure 7. Calculations showed, that three MSRVs were opened for about 200 seconds and about 40 tons of steam were discharged to ACS. Steam through three MSRVs gets into fifth pool of ACS. Water in the fifth pool evaporates in about three minutes. Steam pushes water from 1 - 4 pools and gets to the reinforced compartments of ACS. AZ-1 is activated at the time 260 s due to pressure increase in the ACS compartments. After AZ-1 activation both turbines starts to decrease their throughput down to 150 kg/s. Computed results agree well with the plant data.

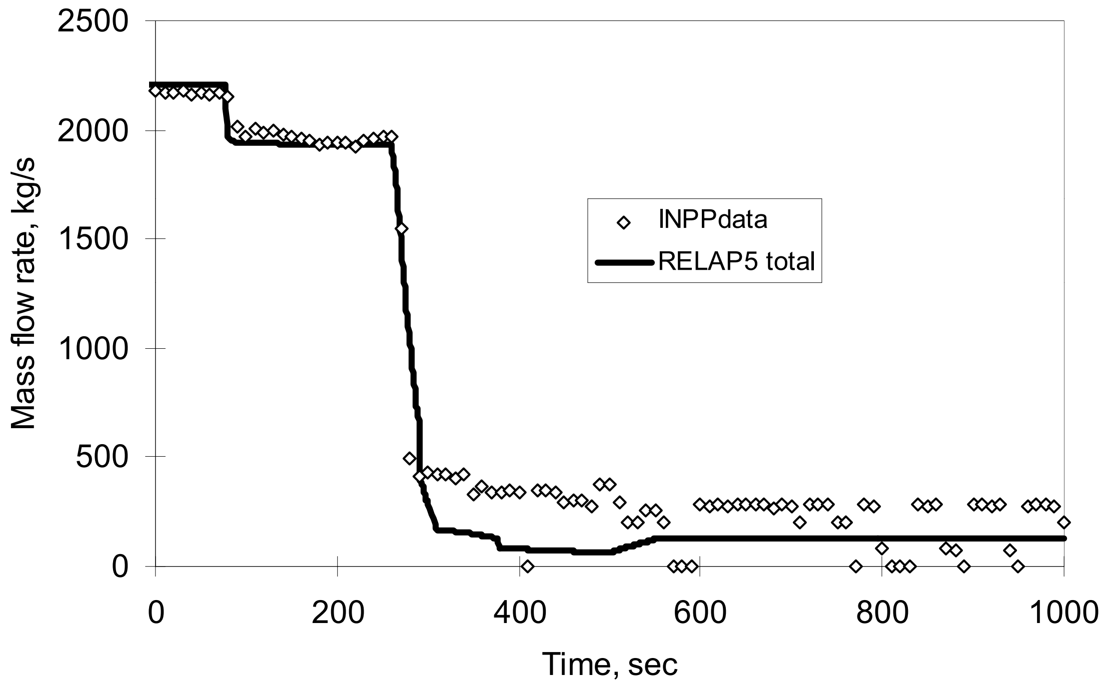

Figure 7. Three MSRVs LOCA event. Total steam flow rate

Inadvertent actuation of ECCS

On the November 22, 1995 fast acting valves, which are used for isolation of ECCS accumulators from MCC, at Unit 1 of the Ignalina NPP were spuriously opened. Before this event reactor operated at thermal power level of 3525 MW. As a result of inadvertent actuation of ECCS cold water flowed to the GDHs of one loop of MCC. At about 220 second from the beginning of the event operator started to decrease reactor power with aim to diminish power irregularity between loops. During the next 650 seconds the reactor power was decreased manually by operator to thermal power of 3350 MW. ECCS water injection of the water takes about 30 seconds. During this period the water level in the ECCS accumulators decreased from 5.60 m to 5.20 m and the pressure decreased from 8.93 MPa to 7.92 MPa. However, in such short term spurious delivery of ECCS water to the primary circuit had a small influence on the main process parameters in the primary circuit, but noticeable changes of the water level in drum separators, steam and feed water flow rates has been observed.

For better presentation of results, it was assumed in calculations that ECCS fast acting valves have spuriously opened at 30th second. After opening of fast acting valves, water from eight ECCS accumulators is supplied to GDH of right loop. The water flow rate from accumulators is shown in Figure 8. Injection of the ECCS water to the right MCC loop takes about 32 seconds. Maximum flow rate is about 700 kg/s. This leads to water level decrease in accumulators in about 0.4 m. Water supply continues until pressure in accumulators and pressure in the GDH become equal. This is the main reason why supply of the ECCS water to primary circuit has been terminated. Results of RELAP5-3D analysis are in good agreement with actual data.

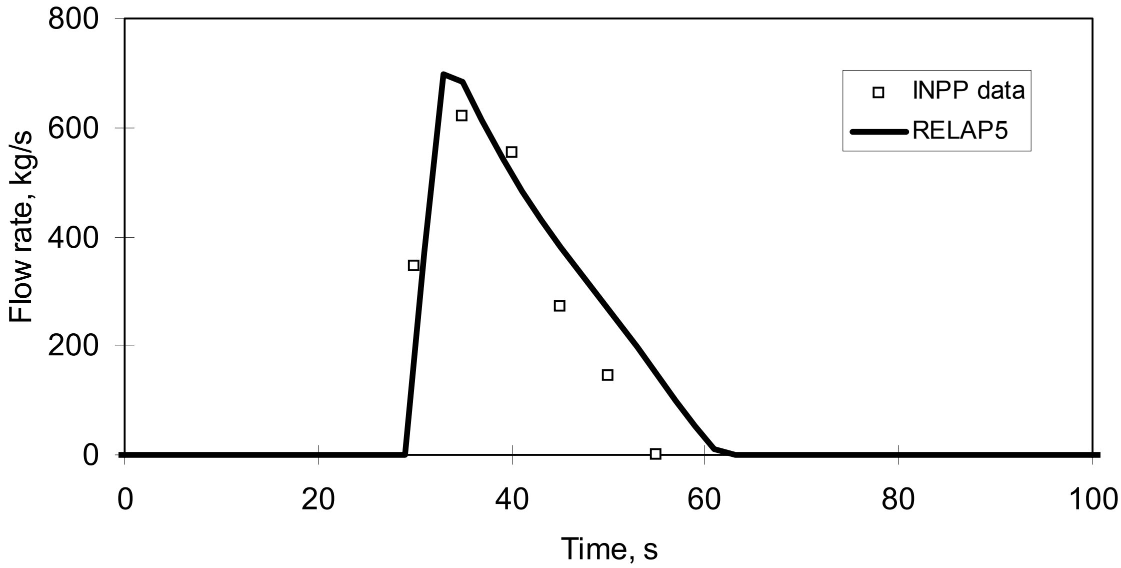

Figure 8. Inadvertent actuation of ECCS. Mass flow from the ECCS accumulators

3.2. Verification of the nodal kinetics part of the RELAP5-3D model using real plant data

First steady-state calculations of the Ignalina NPP RBMK-1500 reactor (2 unit, reactor core state for 1998.11.26.) core state were made and the first calculation results obtained for comparison with the real plant data and the calculation results of the same reactor core state obtained using German neutron-dynamic code QUABOX/CUBBOX. Parameters that were compared are: radial power distribution, axial power distribution, eigenvalue and coolant density profile in fuel channels in the core region.

Figures 9 and 10 show the comparison of the radial power distribution as calculated by codes RELAP5-3D and QUABOX/CUBBOX with the real plant data. In these pictures compared are RELAP5-3D code calculation results (indexes: RELAP x=29 and RELAP y=29), QUABOX/CUBBOX code calculation results, obtained using plant data without any fuel assembly burnup corrections (indexes: Q/C x=24-nc and Q/C y=25-nc), QUABOX/CUBBOX code calculation results, obtained using special fuel assembly burnup correction by 5% to fit the power distribution profile in the core measured by in-core detectors (indexes: Q/C x=24-c and Q/C y=25-c) and the real plant data (indexes: INPP x=24 and INPP y=25). As one can see radial power distribution calculated by RELAP5-3D code is very similar to the one calculated by QUABOX/CUBBOX code, obtained using plant data without any fuel assembly burnup corrections, although RELAP5-3D code gives slightly lower power values. But both these radial power distributions are still lower than the radial power distribution values obtained from the database. In Figure 10 one can notice also two places at the periphery of the core, where RELAP5-3D and QUABOX/CUBBOX values of radial power distribution are higher than the values obtained from the database. Still the biggest deviations from the plant data are in the reactor core center, but going to the periphery of the core the calculated power values agree quite well with the real plant data, except for several points. Now lets compare the radial power distribution values, obtained by QUABOX/CUBBOX code using special fuel assembly burnup correction by 5% to fit the power distribution profile in the core measured by in-core detectors with the plant data and the calculation results, obtained by RELAP5-3D and QUABOX/CUBBOX codes, using plant data without any fuel assembly burnup corrections. Calculation results, obtained by QUABOX/CUBBOX code using fuel assembly burnup correction fits quite well the plant data obtained from the database, and are closer to the reality than the calculation results, obtained by RELAP5-3D and QUABOX/CUBBOX codes using plant data without any fuel assembly burnup corrections. Here one can see the difference in the calculation results which is due to the availability or unavailability of the fuel assembly burnup correction procedure, used during calculations.

Figure 11 shows the comparison of the coolant density axial distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data. As one can see, the calculation results and plant data agree quite well. Some disagreement with the plant data could be seen in QUABOX/CUBBOX results near the bottom of the core and in RELAP5-3D results near the top of the core, but the difference is rather small.

Figure 12 shows the comparison of the axial power distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data. The calculated axial power distribution profile and the measured one at the Ignalina NPP agree reasonably well. Slightly bigger difference is noticed in QUABOX/CUBBOX calculation results, where two power peaks could be seen, one located ~2 m from the top of the core, and another located ~5 m from the top of the core. These are the places where a center of the two fuel bundles in a single fuel assembly are located. On the whole, the axial power non-uniformity value is acceptable being ~1.22. Eigenvalue obtained by RELAP5-3D code is ~1.0013, while eigenvalue obtained by QUABOX/CUBBOX code is ~0.997.

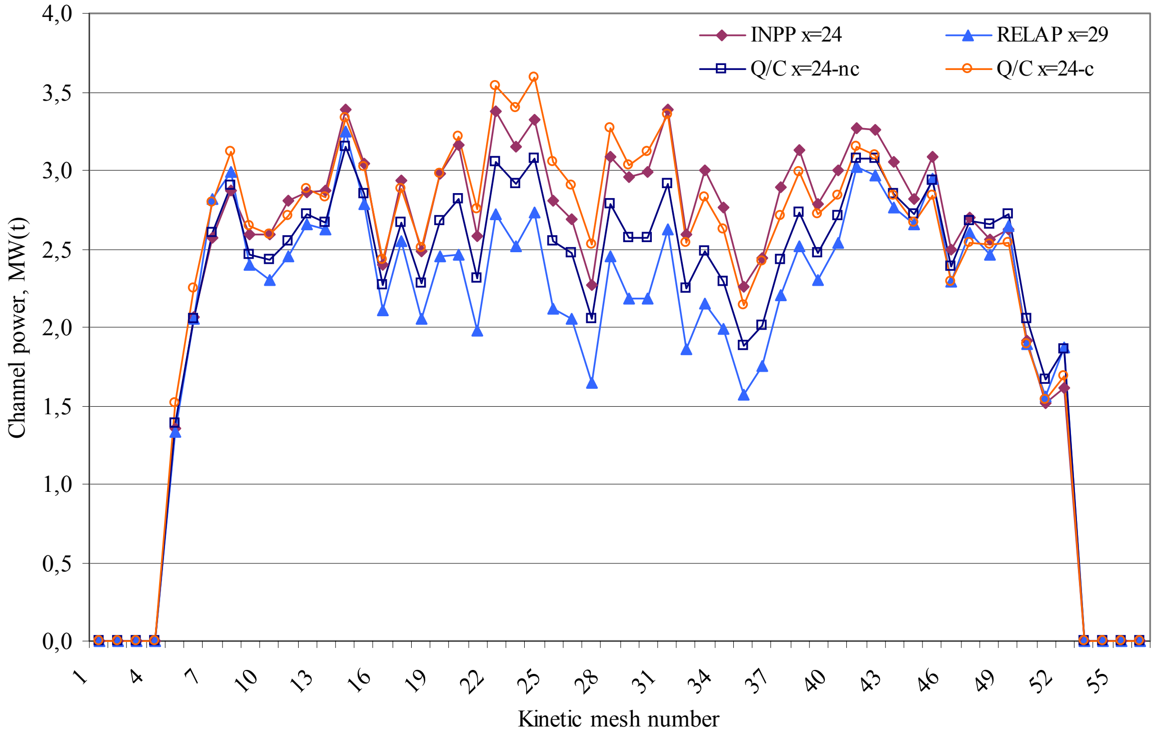

Figure 9. Comparison of the radial power distribution as calculated by codes RELAP5-3D and QUABOX/CUBBOX with the real plant data in X direction in the center of the core

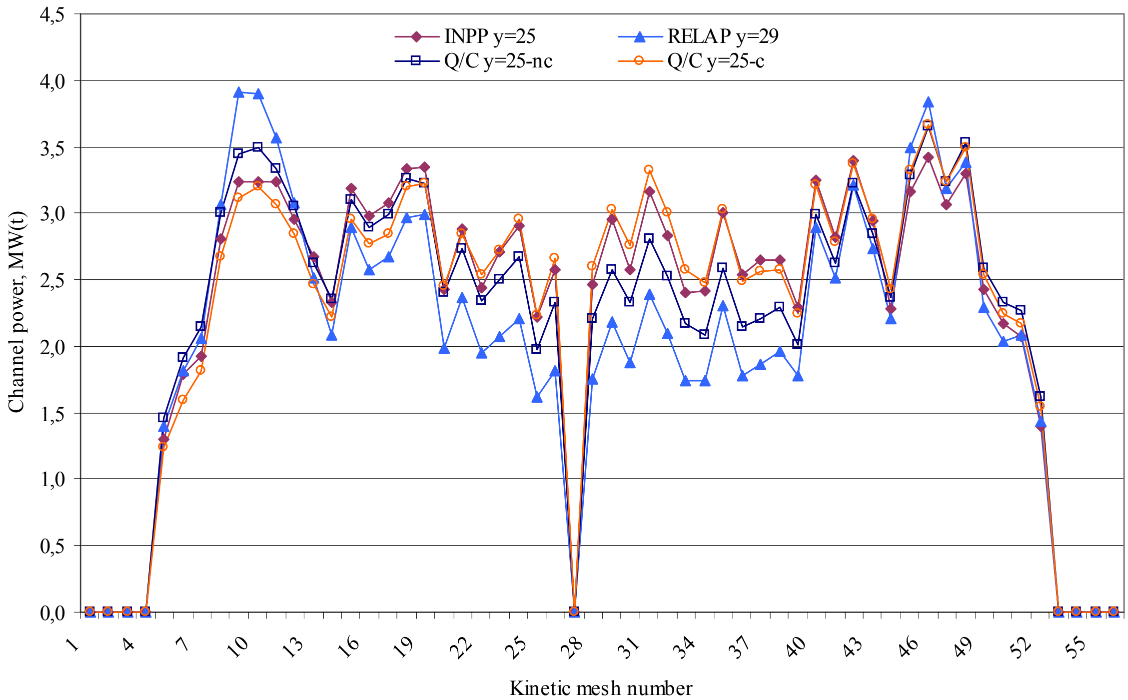

Figure 10. Comparison of the radial power distribution as calculated by codes RELAP5-3D and QUABOX/CUBBOX with the real plant data in Y direction in the center of the core

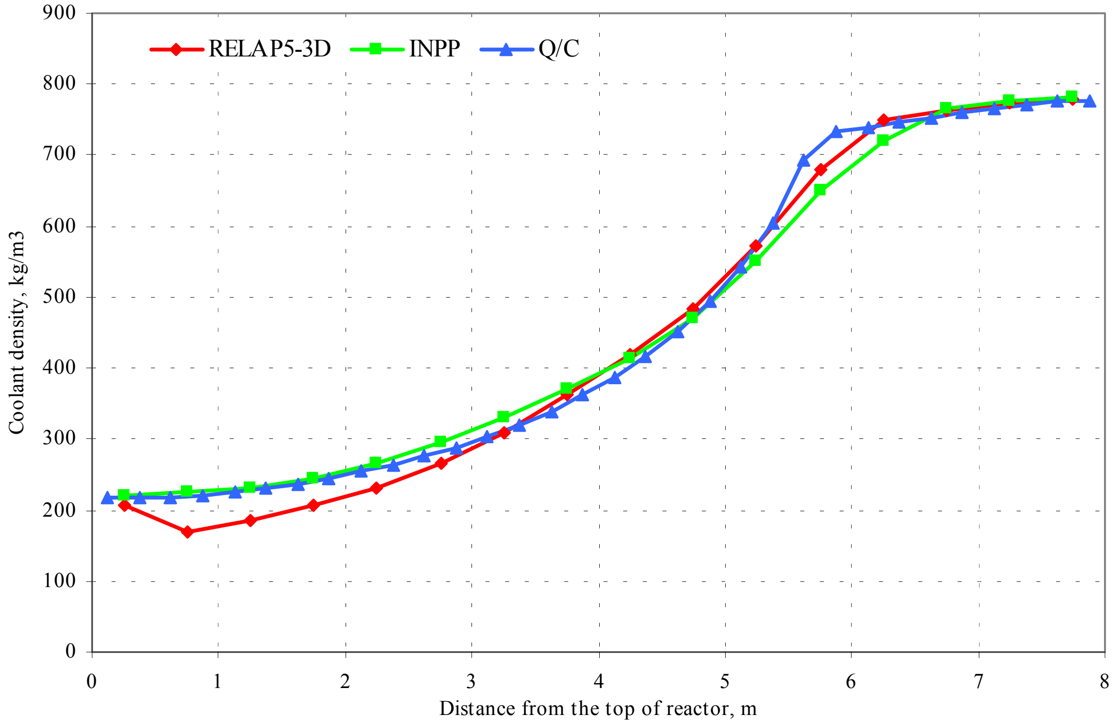

Figure 11. Comparison of the coolant density axial distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data

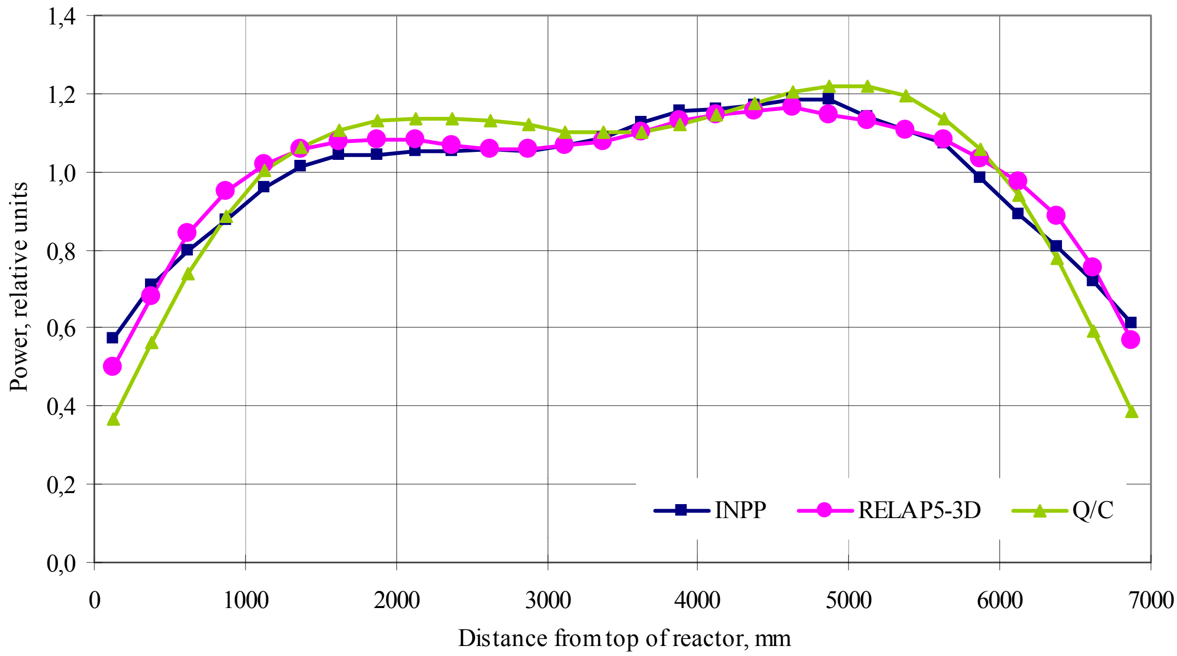

Figure 12. Comparison of the axial power distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data

A conclusion could be made, that the first calculation results of RBMK-1500 reactor core state obtained using RELAP5-3D code agree quite well to the real plant data and the RELAP5-3D nodal kinetics model represents the Ignalina NPP Unit 2 reactor power and coolant density profiles reasonable well. Eigenvalue close to unity indicates reasonable values are calculated for neutron fluxes.

4. Conclusions

1. A successful best estimate RELAP5-3D model of the Ignalina NPP has been developed.
2. The verification of the model has been performed using operational transients from the Ignalina NPP. The results of the calculations obtained with RELAP5-3D model on the Ignalina NPP specific base compare favorably with the plant data.
3. The steady-state calculation results of RBMK-1500 reactor core state obtained using RELAP5-3D code agree well to the real plant data. The RELAP5-3D nodal kinetics model represents the Ignalina NPP Unit 2 reactor power and coolant density profiles reasonable well, too. Eigenvalue close to unity indicates reasonable values are calculated for neutron fluxes.

5. Acknowledgements

The authors would like to acknowledge the technical (from INEEL), financial support and access to the code RELAP5-3D provided by the US DOE, the US International Nuclear Safety Program (INSP). Especially we want to thank two experts from INEEL, Dr. James Fisher and Dr. Paul Bayless, who helped a lot with their advises and sharing their experience. All this made it possible to bring together the analytical methodology developed in the US with the design knowledge of the unique RBMK systems available in Lithuania. The authors would like to acknowledge also the technical support and access to the STEPAN x-section library provided by RRC "Kurchatov Institute. Especially we want to thank RRC "KI" experts Dr. A. Krajushkin, Dr. A. Balygin and Dr. A. Glembotskiy for their advises and their contribution to the successful realization of the first part of the project. We also want to extend our thanks to the administration and technical staff of the Ignalina NPP, for providing information regarding operational procedures and operational data.

6. Abbreviations

ACS -	Accident Confinement System
ANL -	Argonne National Laboratory
AZ-1,3,4,6 -	Emergency Protections 1, 3, 4, 6
CPS -	Control and Protection System
DBA -	Design Basis Accident
DKER -	Russian Acronym for "Power Density (Distribution) Monitoring Sensor Radial
DKEV -	Russian Acronym for "Power Density (Distribution) Monitoring Sensor Axial
DS -	Drum Separator
ECCS -	Emergency Core Cooling System
FASR -	Fast Acting Scram Rod
FASS -	Fast Acting Scram System
FC -	Fuel Channel
GDH -	Group Distribution Header
ICS -	Information Computer System
INEEL -	Idaho National Engineering and Environmental Laboratory
INSP -	International Nuclear Safety Program
LAC -	Local Automatic Control
LEI-	Lithuanian Energy Institute
LEP -	Local Emergency Protection
LOCA -	Loss-of-Coolant Accident
MCC -	Main Circulation Circuit
MCP -	Main Circulation Pump
MCR -	Manual Control Rod
MSRV -	Main Steam Relief Valve
NPP -	Nuclear Power Plant
PH -	Pressure Header
PNNL -	Pacific Northwest National Laboratory
RBMK -	Large Channel Type Water Cooled Graphite Moderated Reactor
RDIPE -	Research and Development Institute of Power Engineering
RRC "KI" -	Russian Research Center "Kurchatov Institute"
SACR -	Short Absorber Control Rod
SH -	Suction Header
T-H -	Thermal-Hydraulic
US DOE -	Department of Energy of the United States of America

7. References

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Development of Ignalina NPP RBMK-1500 reactor RELAP5-3D model

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Project

In 1999 a contract was signed between the Pacific Northwest National Laboratory (PNNL), which is representing US DOE in the frame of the US International nuclear safety program (INSP) and, as a part of it, the Soviet-designed reactor safety program, and the Lithuanian Energy Institute (LEI).

Project Goals

The overall objective is to develop models for RBMK-1500 reactors for use in integrated thermal-hydraulics-neutronics calculations for the analysis of specific transients in which the neutronic response of the core is important.

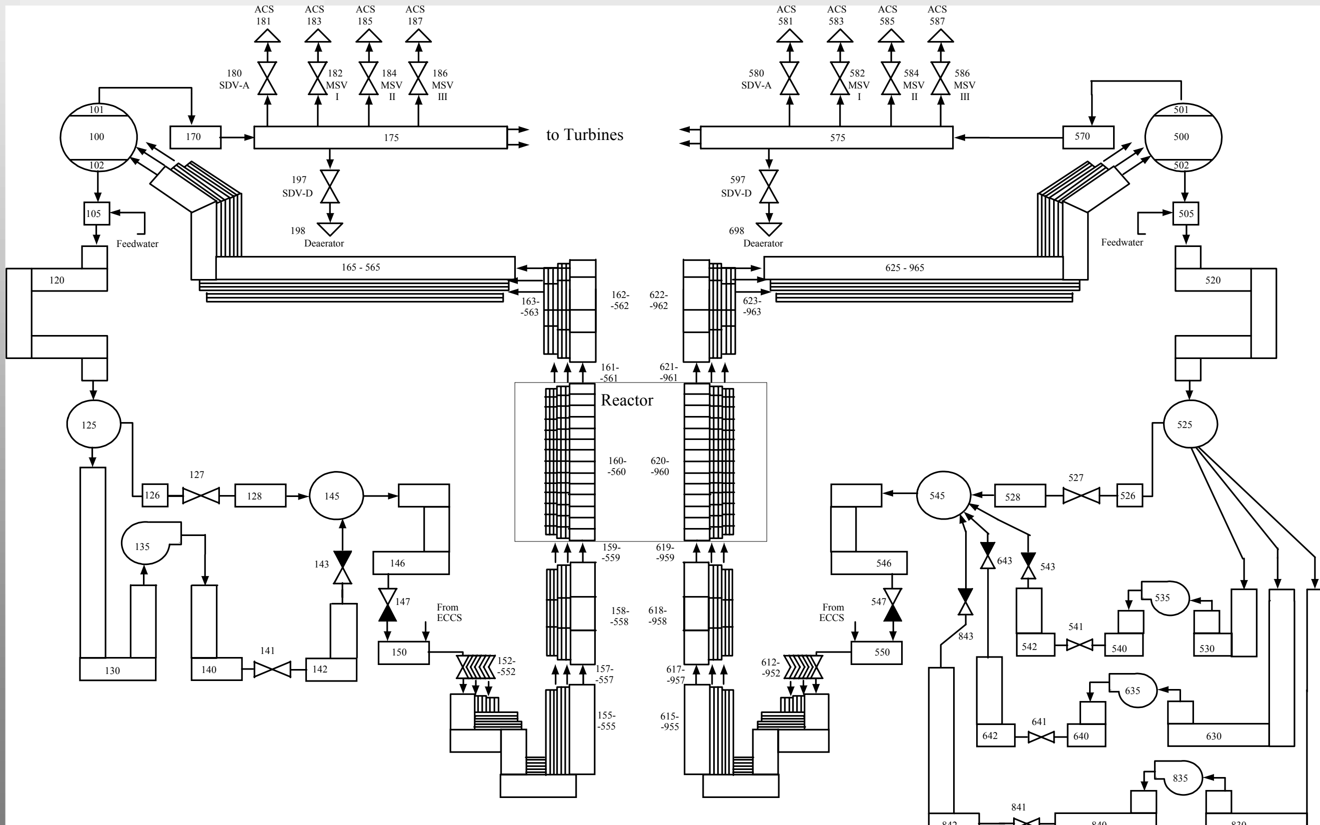
The specific technical goals:

- development of the neutron kinetics and thermal-hydraulics RELAP5-3D model;
- verification of the coupled code RELAP5-3D model using real plant data;
- analysis of the defined benchmark problems.

Key features of Ignalina NPP RELAP5-3D model

- **Modeled are: 1) both MCC loops; 2) paths for steam removal from DS; 3) feedwater and ECCS systems; 4) automatic regulators of pressure and water level in DS; 5) control rod and radial reflector cooling circuit; 6) CPS logic;**
- **The heat exchange between technological channels is modeled using heat exchange through the gap in graphite moderator in a gas circuit of the reactor. The transfer of heat from FC to CPS channels and reflector cooling circuit occur through the same gas circuit;**
- **The reactor thermal power is calculated using the nodal kinetic model of RELAP5-3D code.**

Thermal-hydraulic part of INPP RELAP5-3D model

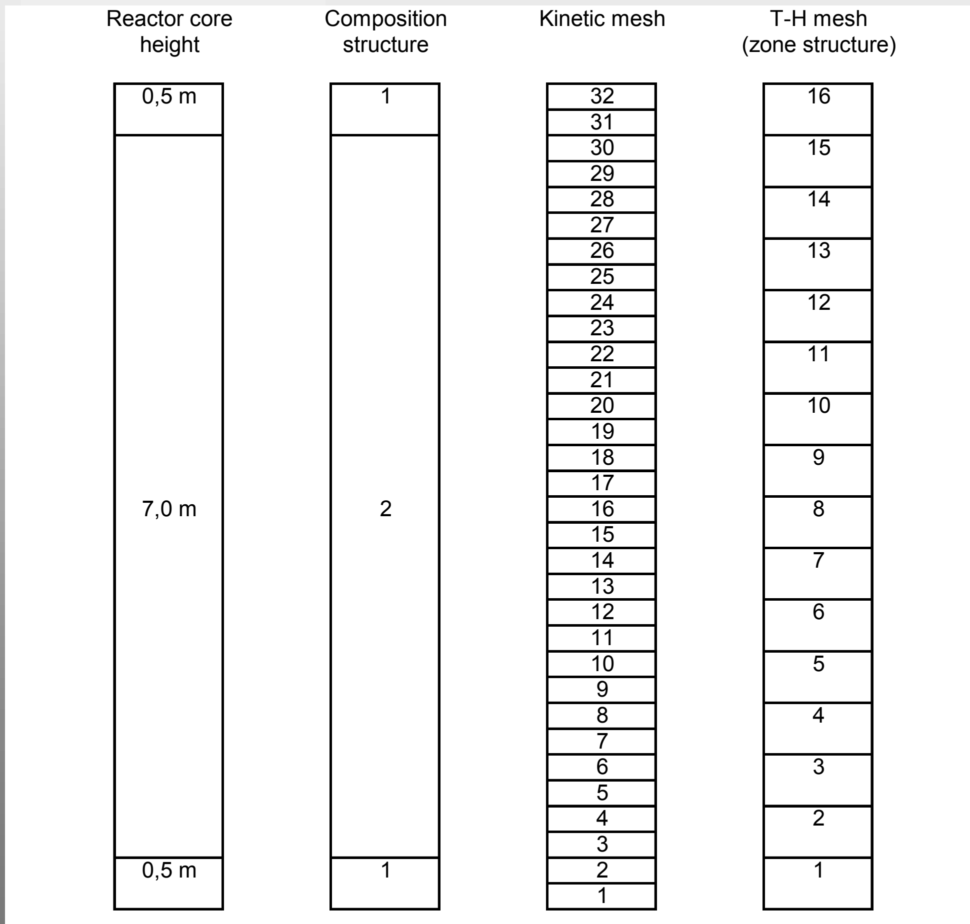


- **Model of the MCC consist of two loops: left half is simplified, but right half is modeled in finer detail;**

Thermal-hydraulic part of INPP RELAP5-3D model, cont.

- **Reactor core is modeled by 14 pipe components: 7 represent 835 FC of the left loop and the rest 7 represent 826 FC of the right loop;**
- **Each equivalent FC is modeled using 16 axial nodes (0.5 m each);**
- **FC and graphite columns are modeled using 8 radial nodes: 2 for FC wall, 2 for gap and graphite rings region and 4 for graphite column;**
- **Fuel element is modeled using 8 radial nodes: 5 to represent fuel pallet, 1 for the gap region and 2 for cladding.**

Nodal kinetics part of INPP RELAP5-3D model



- **RBMK-1500 reactor core has 7.0 m fuel region and 0.5 m reflector region above and below the fuel region;**
- **Neutronics mesh represent each rectangular graphite column as one individual stack in the radial plane;**
- **Reactor core region has**

Nodal kinetics part of INPP RELAP5-3D model, cont.

- Reactor core composition is represented by 2 composition maps. 12 different compositions are present in the reactor core layout;
- Nodal kinetics model is based on the real state of reactor of INPP Unit 2 on November 26, 1998;
- Database information used: core loading, burnup of each FA, axial and radial fuel burnup profile, coolant flowrate information in MCC and CPS cooling channels, insertion depth of CPS CRs;
- X-sections for different compositions of the reactor core were obtained from two-group macro x-section library of the STEPAN code;

Nodal kinetics part of INPP RELAP5-3D model, cont.

- **External user subroutine interface was written that accesses the coding of x-section library at each step of calculation. Thermal-hydraulic and CR position information being the input to the interface, while diffusion, absorption, fission and scattering x-sections for two neutron groups being the output of the interface;**
- **For fuel cells x-section library needs relative node power level to correct the Xe radial and axial distribution in the core. For the first steady-state run the relative power is set equal to 0.864 ($N_{actual}/N_{max}^{design}$) for all kinetics nodes;**

Nodal kinetics part of INPP RELAP5-3D model, cont.

- **For the next steady-state run steps it is calculated using the equation:**

pow = (phi(1)*sigf1p(ixyz) + phi(2)*sigf2p(ixyz))*G*K*V*N/R,

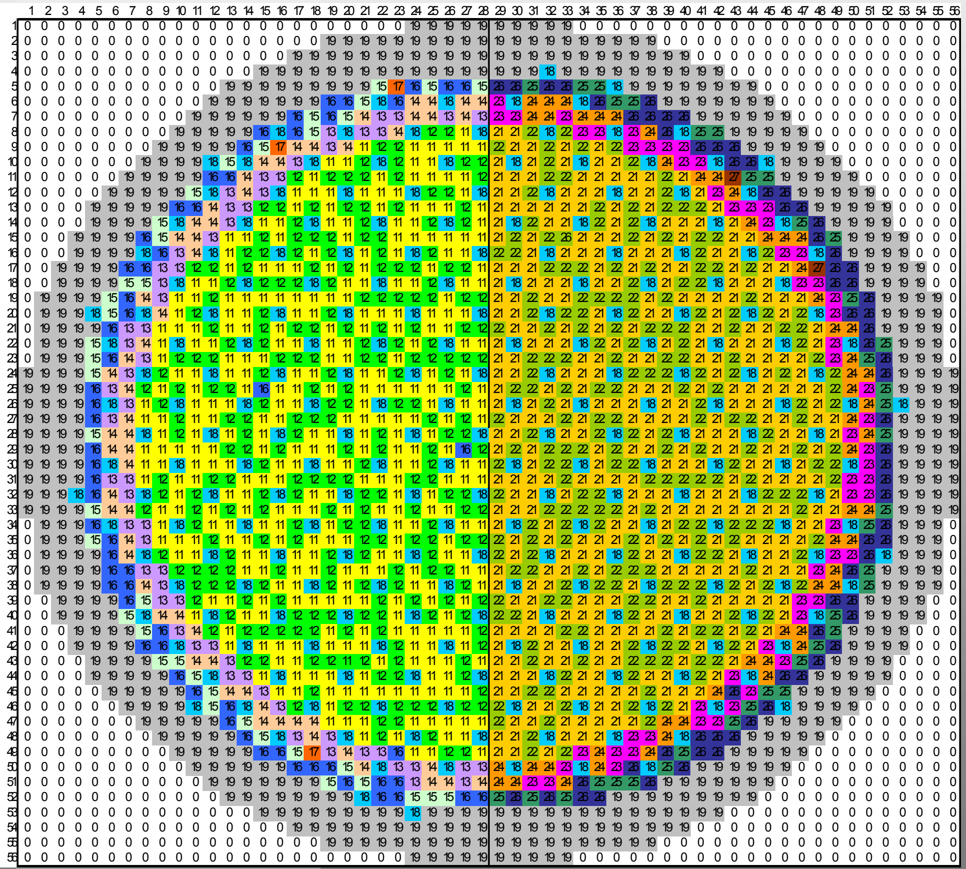
where: phi(i) - the neutron flux for group i in the current mesh position; sigf1p(ixyz) - the macroscopic fission x-section for group i for current mesh position (saved from previous time step); G = 200 MeV/fission; K = 1.6021917e-13 J/MeV; N = 1661 - number of fuel assemblies; V = 25*25*700 = 437500 cm^3 – fuel assembly volume; R = 4800 MW - rated power;

- **This equation is used because flux values are available directly to external subroutine, whereas node power is not;**

Nodal kinetics part of INPP RELAP5-3D model, cont.

- **Relative node power value is used in steady-state case to obtain xenon relative equilibrium value;**
- **For transient case this value is ‘frozen’ and taken as being constant for each kinetics node;**
- **Reactor core is divided into 2 halves: 7 T-H channels for FCs per core half and 2 T-H channels for non-fuel channels (CPS and RRCC). FCs are divided into 7 groups according to FA power and FC coolant flowrate values;**
- **Kinetics part of the model models each fuel and non-fuel channel individually;**

Nodal kinetics part of INPP RELAP5-3D model, cont.



annel group ecification in he kinetics model	Channel group specification in thermal-hydraulic model	Side of the reactor	Number of channels	Average power value in a single channel, MW	Average flowrate value in a single channel, m^3/h
CC11	620	Left	355	2.95	28.2
CC21	160	Right	378	2.95	28.2
CC12	710	Left	249	2.5	26.2
CC22	210	Right	234	2.5	26.2
CC13	730	Left	60	2.4	25.1
CC23	280	Right	59	2.4	25.1
CC14	760	Left	59	1.8	21.1
CC24	390	Right	55	1.8	21.1
CC15	780	Left	39	1.6	17.5
CC25	410	Right	37	1.6	17.5
CC16	860	Left	61	1.2	15.6
CC26	460	Right	70	1.2	15.6
CC17	960	Left	3	1.8	33.5
CC27	560	Right	2	1.8	33.5
CC18	022		235		
CC19	032 & 034		592*		

* 436 channels are radial reflector channels

Nodal kinetics part of INPP RELAP5-3D model, cont.

- **Another complicated part of the model is CPS CRs and CPS operation logic;**
- **All 211 CPS CRs (4 types) are modeled individually, because of different axial positions;**
- **RELAP5-3D control variable system is used for CPS logic and CPS CR movement modeling;**
- **Movements of CPS CRs are controlled by CPS logic, based on power deviation signals coming from 127 radial detectors of DKER-1detector system, which are modeled as having 7 sensitive elements (0.25 m each) distributed evenly over the height of the fuel region of the reactor core;**

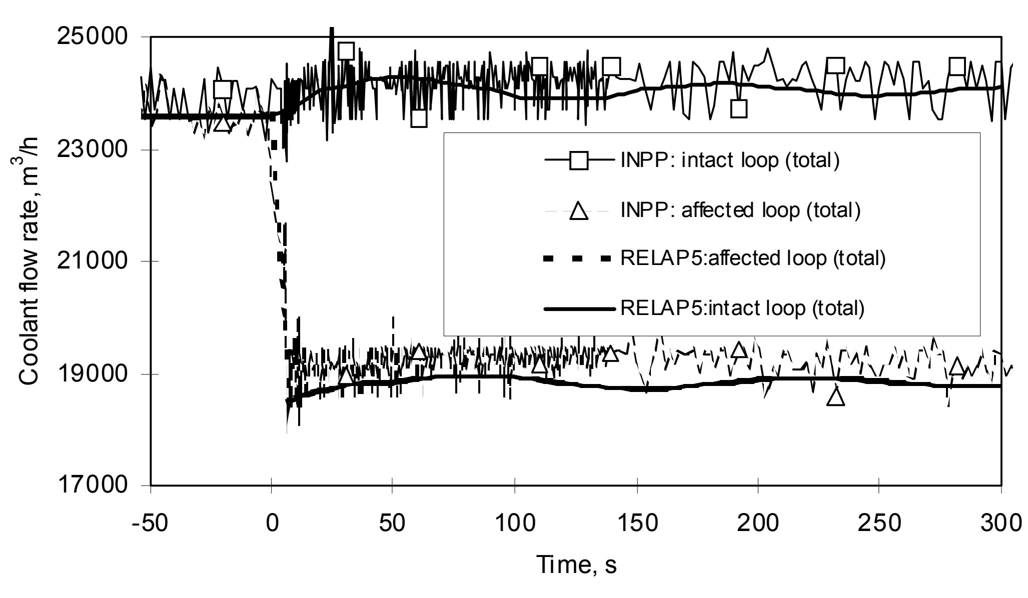
Nodal kinetics part of INPP RELAP5-3D model, cont.

- Power deviation signal is based on the steady-state thermal neutron flux value in each detector location;
- All DKER-1 detectors are located in 12 LAC/LEP zones, that have 1 LAC and 2 LEP CRs each;
- The following emergency protection modes are modeled: AZ-1, FASS, AZ-3 (power reduction to 0.5Nn), AZ-4 (power reduction to 0.6Nn) and AZ-6;
- 125 DKER-2 detectors are not modeled because they are mainly used for technological purposes;
- 160 DKEV detectors still need to be implemented if control variable system will not be a limitation.

Verification of thermal-hydraulic part of INPP RELAP5-3D model

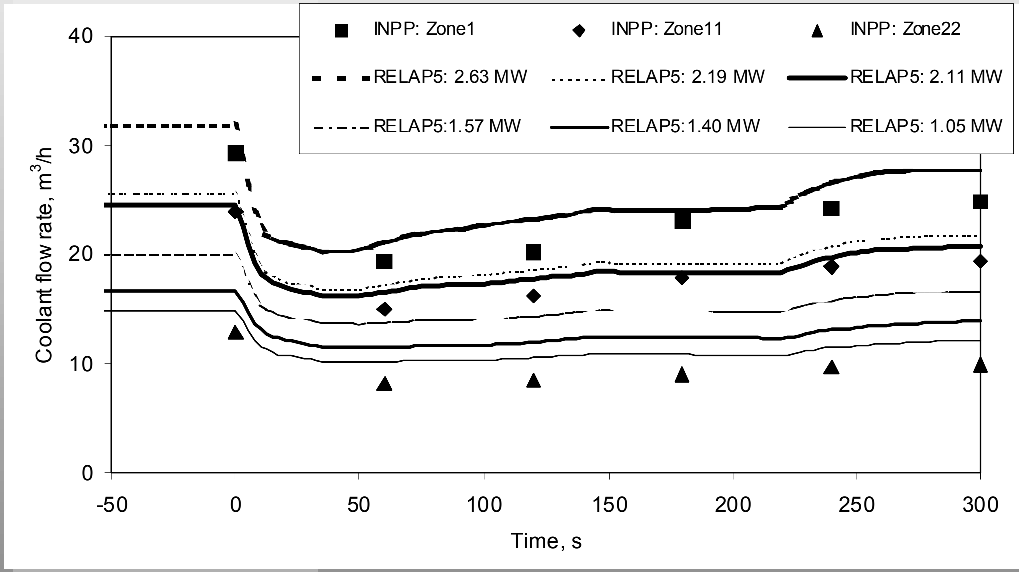
- **Comparison of calculation results with real plant data allow to verify the suitability of the developed model for future modeling of processes, taking place in RBMK-1500 reactor;**
- **For this purpose the following benchmark analysis were performed:**
  - **Single MCP trip,**
  - **One MCP trip with failure of check valve,**
  - **Loss-of-all-MCPs transient,**
  - **Three MSRVs LOCA event,**
  - **Inadvertent actuation of ECCS.**

Verification of T-H part of INPP RELAP5-3D model, cont.



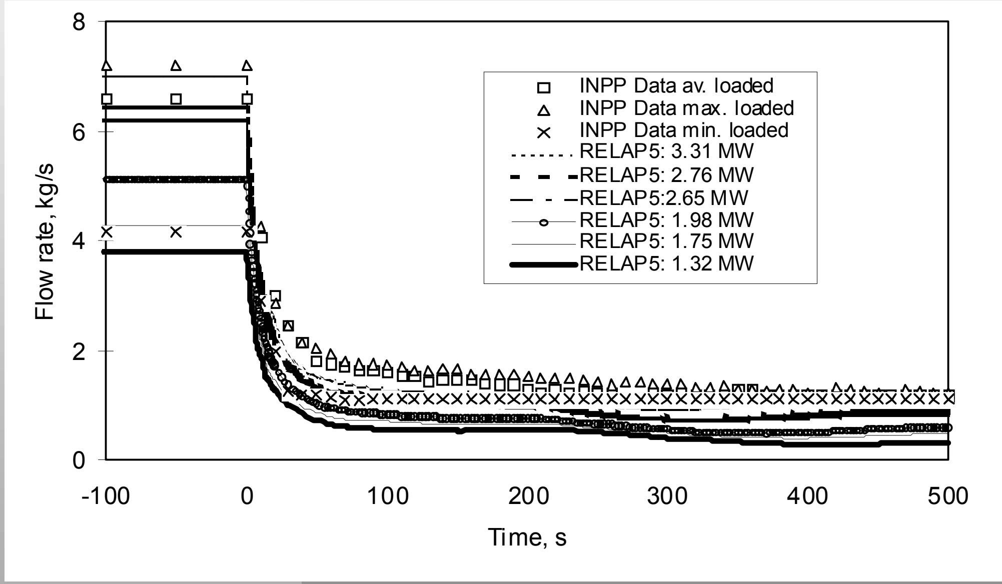
- **On May 14, 1996 one MCP at Ignalina Unit 2 was inadvertently tripped. The reactor prior to the event was operated at 3400 MW(th) power. AZ-4 signal was generated due to loss of power to the MCP. After one MCP trip, the throughput of two running pumps increased by ~1500 m^3/h. However, the total coolant flow through the affected loop decreased from 23500 m^3/h to 19000 m^3/h. The figure shows a favorable coincidence of MCP throughputs and coolant flow rate through the reactor.**

Verification of T-H part of INPP RELAP5-3D model, cont.



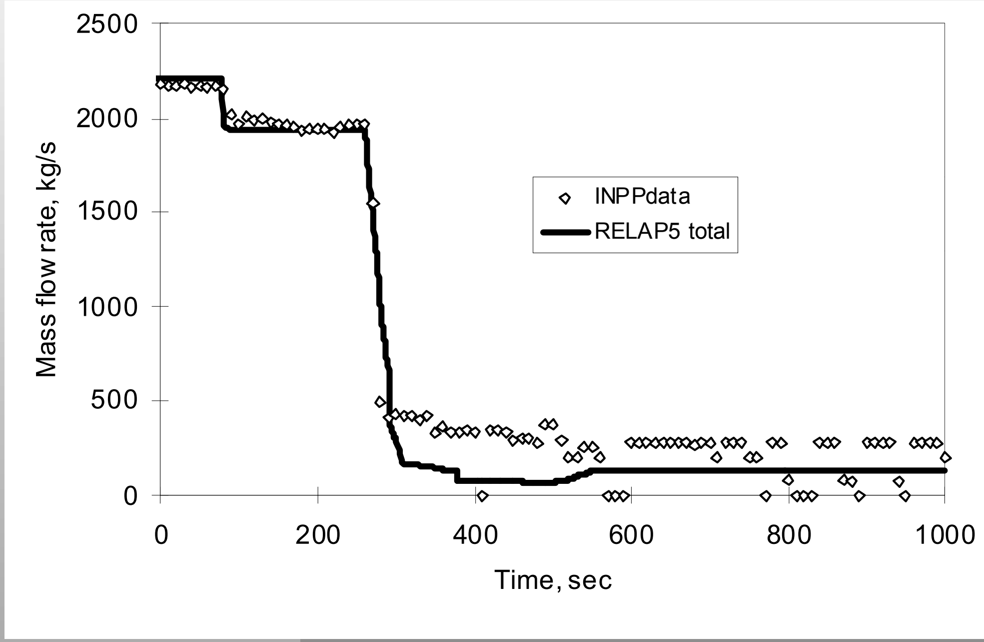
- The similar event took place on January 23, 1998. In this case, one MCP trip with failure of check valve occurred. The reactor prior to the event was operated at 3700 MW(th) power. After the pump trip each of the two operating pumps (in the affected loop) increased its throughput from 7750 m^3/h to 10100 m^3/h. The net flow supplied to the affected core side decreased from 2300 m^3/h to 15500 m^3/h. Measured data and calculated flows through fuel channels agree quite well.

Verification of T-H part of INPP RELAP5-3D model, cont.



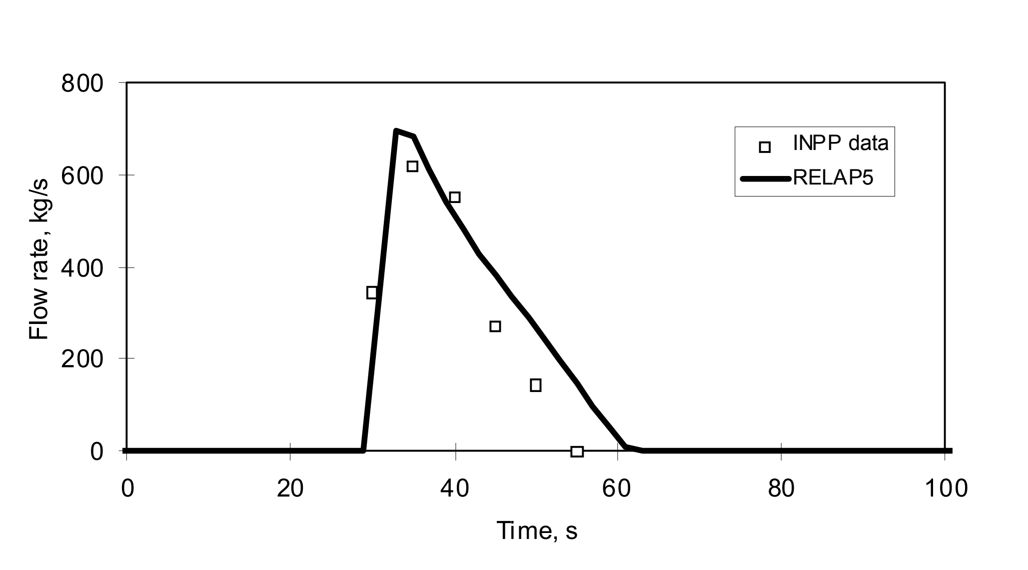
- This is an actual transient that occurred at Ignalina NPP on March 26, 1986 - all six operating MCPs at Ignalina Unit 1 were tripped simultaneously. Before this event reactor was operated at 4650 MW(th) power level. In response to multiple pump trip, an emergency protection signal AZ-1 was generated and reactor was shutdown. The MCC flow decreased in response to the MCPs cost-down. Flow rate through the fuel channel decreased due to loss of forced circulation by the MCPs. The analysis results agree well with the measured flow rates.

Verification of T-H part of INPP RELAP5-3D model, cont.



- 3 MSRVs of 3rd group were spuriously opened at INPP Unit 1 on November 27, 1986. 3 MSRVs spuriously opened on 79 s and closed on 261.5 s from beginning of the accident. Here you see total steam flow rate to turbines and for local consumers. Steam through 3 MSRVs gets into 5th pool of ACS. Water in 5th pool evaporates in about 3 min. Steam pushes water from 1-4 pools and gets to reinforced compartments of ACS. AZ-1 is activated at time 260 s due to pressure increase in ACS reinforced compartments. After AZ-1 activation both turbines starts to decrease their throughput down to 150 kg/s. Computed results agree well with the plant data.

Verification of T-H part of INPP RELAP5-3D model, cont.

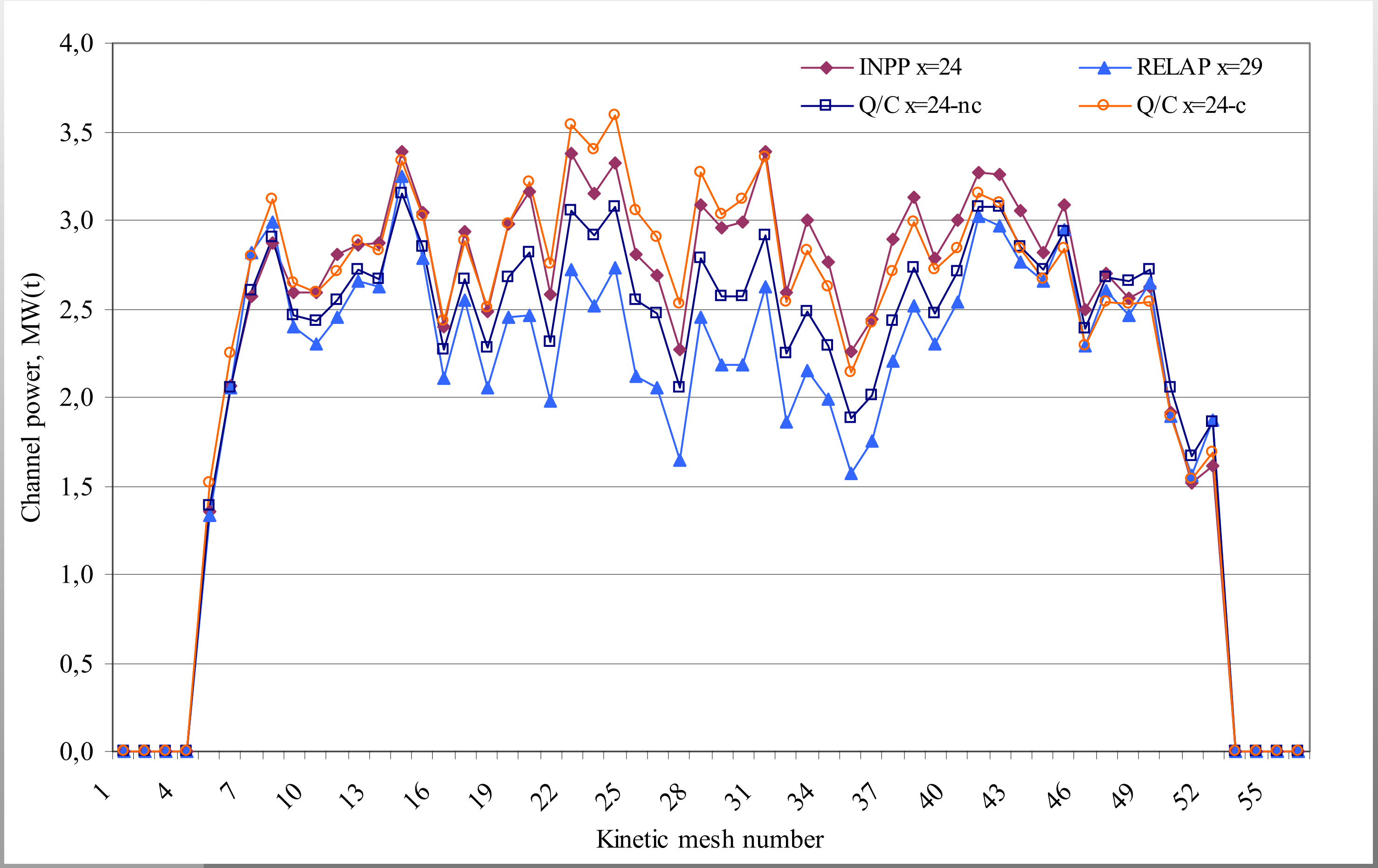


- On November 22, 1995 fast acting valves, which are used for isolation of ECCS accumulators from MCC, at Ignalina NPP Unit 1 were spuriously opened. Before this event reactor operated at 3525 MW(th) power level. After opening of fast acting valves, water from eight ECCS accumulators is supplied to GDH of the right loop. Injection of the ECCS water to the right MCC loop takes about 32 s. Maximum flow rate is about 700 kg/s. Water supply continues until pressure in accumulators and pressure in the GDH become equal. Results of RELAP5-3D analysis are in good agreement with the actual data.

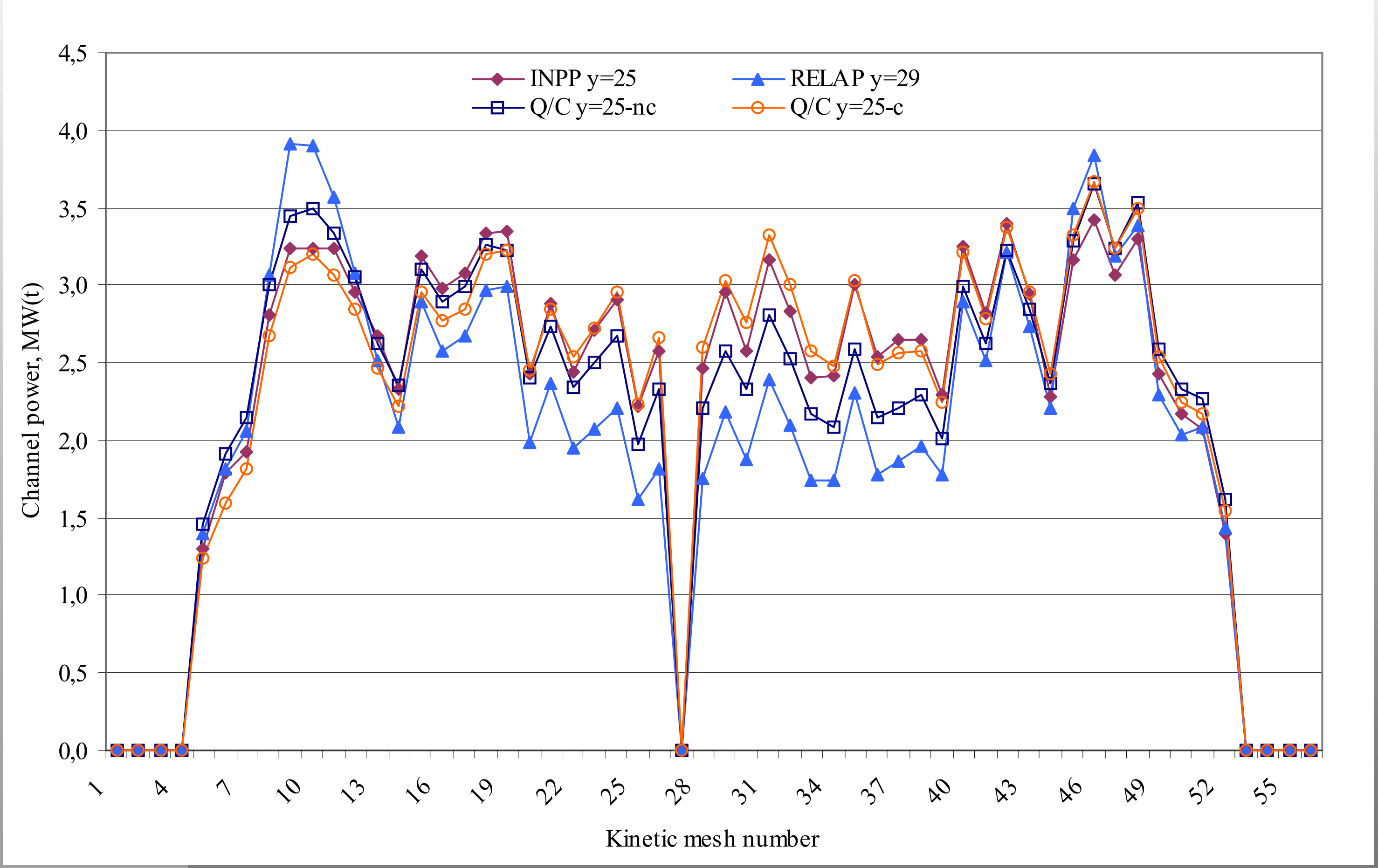
Verification of nodal kinetics part of INPP RELAP5-3D model

- **First steady-state calculations of the Ignalina NPP RBMK-1500 reactor (2 unit, reactor core state for 1998.11.26.) core state were made and the first calculation results obtained for comparison with the real plant data and the calculation results of the same reactor core state obtained using German neutron-dynamic code QUABOX/CUBBOX. Parameters that were compared are: radial power distribution, axial power distribution, eigenvalue and coolant density profile in fuel channels in the core region;**
- **Eigenvalue obtained by RELAP5-3D code is ~1.0013, while eigenvalue obtained by QUABOX/CUBBOX code is ~0.997.**

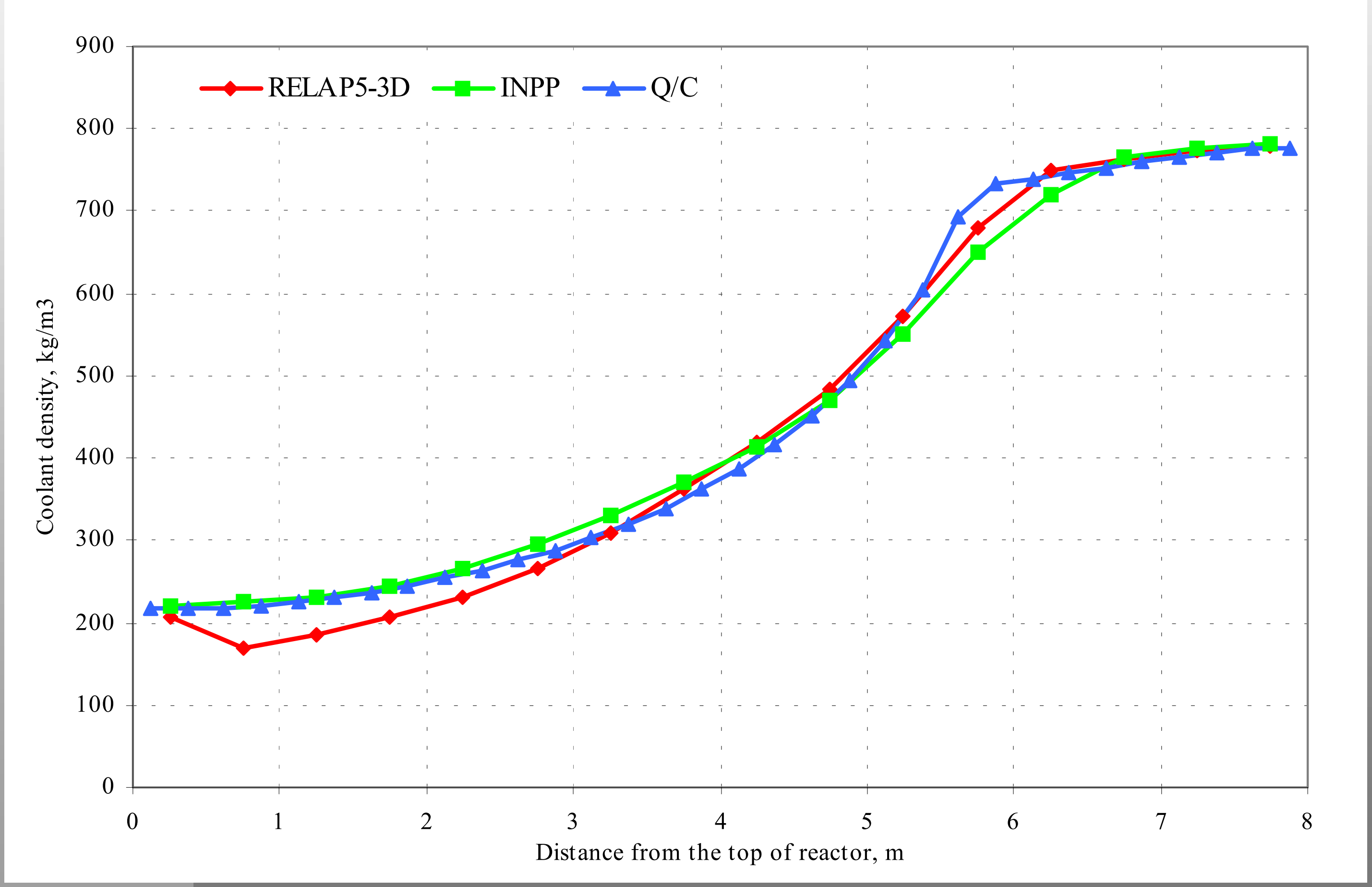
Verif. of nodal kinetics part of INPP RELAP5-3D model, cont.



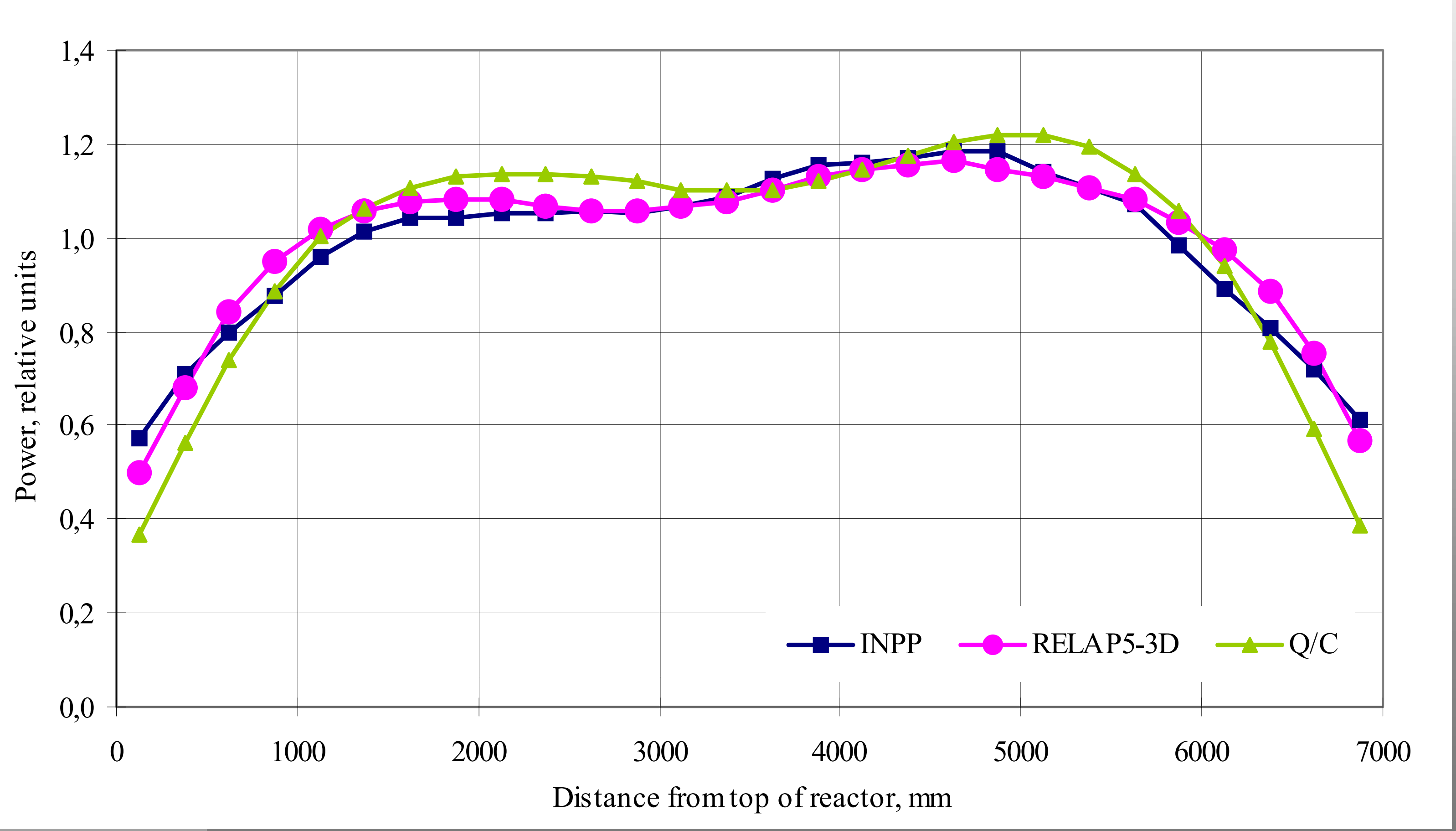
Verif. of nodal kinetics part of INPP RELAP5-3D model, cont.



Verif. of nodal kinetics part of INPP RELAP5-3D model, cont.



Verif. of nodal kinetics part of INPP RELAP5-3D model, cont.



Conclusions

- A successful best estimate RELAP5-3D model of the Ignalina NPP has been developed;
- The verification of the model has been performed using operational transients from the Ignalina NPP. The results of the calculations obtained with RELAP5-3D model on the Ignalina NPP specific base compare favorably with the plant data;
- The steady-state calculation results of RBMK-1500 reactor core state obtained using RELAP5-3D code agree well to the real plant data. The RELAP5-3D nodal kinetics model represents the Ignalina NPP Unit 2 reactor power and coolant density profiles reasonable well, too. Eigenvalue close to unity indicates reasonable values are calculated for neutron fluxes.

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